## Project Title

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**Abstract**

This report involves simulating a four loop Pressurized Water Reactor (PWR) with a single pass core using a numerical loop momentum balancing approach. The first section details the theoretical basis for the pressurized water reactor and how the momentum balancing equations provide an accurate approximation for the reactor physics. Next, the report dives into the specific methods used in order to discretize the momentum balance equations for the specific four loop PWR before determining the appropriate Reactor Coolant Pump (RCP) and Steam Generator design parameters needed to meet required mass flux and core inlet temperature rises. This reactor is then analyzed for performance in three pump transients. The first transient evaluates performance in a loss of all AC casualty where all RCPs trip off simultaneously and coast down. The second evaluates a single RCP trip that coasts down. The third evaluates a locked rotor casualty impacting a single loop, where the loop effectively loses all flow. The fourth transient analyzes a single RCP shear inducing reverse flow through the associated loop. Finally, the number of clogged steam generator tubes are adjusted to determine the proportion required to reduce mass flow rates by 10% of steady state. Reactor safety was demonstrated for the first three transients. The model was unable to successfully demonstrate acceptable performance for the RCP shaft shear condition as it could not produce reverse flow in the core. Future work on this model should expand the reactor safety analysis to observe the impacts of long term decay heat removal and two phase flow.

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# Introduction and Theory

This report analyses many reactor safety considerations which are involved in designing a pressurized water reactor (PWR). PWRs are common in the nuclear power industry, so analyzing this design provides insight into limiting casualties, reactor safety concerns, and the importance of proper reactor plant design. The specific geometry analyzed in this report involves a four loop single pass PWR, where each loop has a Reactor Coolant Pump (RCP), a steam generator, and no check valves to prevent backwards flow. Loop one is used to determine the transient behavior, and the other three loops are grouped into a set of symmetric loops collectively referred to as loop two. A nodal depiction of the PWR is seen in the figure below:

A diagram of a building

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Figure 1: Visualization of PWR Nodes

Where the node categorization is given by the following table:

A table with text and numbers

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Before discussing the results of the transients performed as part of this report, it is important to understand the theory used to numerically model the PWR. Fundamentally, this report calculates the mass flow rate by ensuring momentum is conserved within the reactor. As the heat transfer from both the steam generator and fuel rods depend on the mass flow rate, the flow rate is used to determine the current fluid internal energy and temperatures in the PWR loops and core. While these equations are continuous, this report discretizes the momentum and internal energy at each node shown in Figure 1, which assumes homogenous properties in each node. This report sizes the RCPs and Steam Generators to meet minimum required mass flow rates and core inlet temperatures. Afterward, it analyzes the impact of various transients important for reactor safety on overall reactor performance. The first transient evaluates performance in a loss of all AC casualty where all RCPs trip off simultaneously and coast down. The second evaluates a single RCP trip that coasts down. The third evaluates a locked rotor casualty impacting a single loop, where the loop effectively loses all flow. The fourth transient analyzes a single RCP shear inducing reverse flow through the associated loop. The final transient analyzes the number of clogged steam generator tubes required to reduce mass flow rates by 10%. The objective is to determine the performance of the numerically modeled PWR in these transients after determining the associated design parameters.

# Methods

Momentum Conservation

As momentum is conserved within the reactor, any momentum going into a node must either exit the node or be stored within it. This report models the momentum in each node using the below linearized equations:

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In the above equations, m is the mass flow rate, L is the length of the node, A is the node’s cross-sectional area, D is the node’s equivalent diameter, k is the minor friction losses, f is the Colebrook friction factor, ρ is the water density, and ∆H is the change in height over the node. As the simulation time advances, this momentum balance equation is solved for the node’s mass flow rate at time t+∆t by solving the below matrix equation.

A number and a number of numbers

Description automatically generated with medium confidence

This is implemented in the code below, where the step\_massflux function takes in the model parameters, loop nodes, and core nodes and then iterates through the above momentum equations to solve for the new mass flux values and pressure drop across the core. Each loop and core object have their own momentum method to determine the a and b coefficients:

A computer screen with white text

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A computer screen shot of a code

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After the mass flow rate has been updated for the new time step, the corresponding pump differential pressure is calculated through the following pump curve:

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Internal Energy

As the mass flow rate in each node changes to conserve momentum, the internal energy and corresponding bulk fluid temperature will also change. This behavior is modeled by determining the new equilibrium internal energy at each node with the new equilibrium mass flow rates through the below equation:

A math equations and formulas

Description automatically generated with medium confidence

Here, V is the node volume, q is the heat flux into the fluid from the node (positive for heat flowing into the fluid), and u is the node’s internal energy. There are several methods which can be used to solve the momentum and internal energy equation. In developing the numerical simulation two different methods were investigated – a Newton-Raphson solver and a direct iterative solver. The Newton-Raphson solver takes in a matrix of equations, determines the associated Jacobian matrix, and then solves for new minimum input values based on the derivatives calculated in the Jacobian. The direct iterative solver updates individual inputs based on information from the previously calculated inputs. This process repeats until the individual inputs converge on the steady state values. Below is the implementation of the Newton-Raphson method.

A screen shot of a computer program

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However, this method was not as well behaved as the direct iterative implementation. The direct iterative method solves the internal energy equation directly for the new internal energy at time t+∆t as seen below:

A black and white math symbols

Description automatically generated

This equation was solved continually by updating the internal energy at each time step based on the incoming mass flux and internal energy until a steady state was reached. This approach eliminated the need to solve all internal energy equations simultaneously. In order to reach convergence with small internal energy differences as a result of large mass flow rates, the internal energy was round to four decimal places after each iteration. This project used the direct iterative approach to solve for the internal energy at each node after each new mass flow rate.

Peak Centerline Temperature

In order to determine the maximum surface temperature of the cladding, one dimensional PCT equations were solved based on the current heat flux from the corresponding core node. The Weisman correlation was used to determine appropriate heat transfer for a square lattice of fuel rods:

A black text on a white background

Description automatically generated

As the assignment required maintaining the maximum surface temperature of the fuel cladding below saturation temperature, only convective heat transfer from single phase flow conditions was analyzed. Based on the heat transfer coefficient between the bulk fluid and the fuel rod, the temperature rise at the surface of the cladding is determined through the below equation and python implementation.

A screen shot of a computer code

Description automatically generated

Once the reactor is scrammed, then the reactor power is modeled using the Wigner Way formula:

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Description automatically generated

Putting all of the above equations together, the below script was used to advance the run through time steps based on the provided run time (run\_secs). In order to minimize the amount of time required to run the simulation and reach steady state, after finding the steady state condition the state was saved and reloaded for all future transient runs. The steady state condition ran for 13 seconds before transients were inserted into the reactor, and so for all subsequent graphs the transients occur at t=13 seconds and the simulation ran for five seconds following the transient.

A computer screen shot of a program code

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# Problem Formulation

Scenario A

This scenario sought to determine the necessary RCP ∆P to meet the provided mass flow rates by adjusting the ∆P Rated value. To determine the necessary values, a guess-and-check approach was taken in order to satisfy the minimum plant flowrate and target core inlet temperature.

Scenario B

To run this transient, all four RCPs were set to trip off after steady state operation. This simulates an electrical casualty which immediately trips off all RCPs. After the pumps trip, they begin to coast down to provide 0 ∆P based on the following formula:

A mathematical equation with black text

Description automatically generated

Where β is dictated by the fly wheel inertia of the pump, and t is the time since the pump trip. For a fly wheel with a high inertia, the pump will take longer to coast down which is signified by a larger β value. After the pumps trip, we assume that the automatic protective system takes two seconds to recognize the loss of flow casualty and scram the plant. This scenario seeks to size the pump such that the associated β will prevent thermal limit violations in this transient.

Scenario C

This transient is a less severe instance of Scenario B, as only one RCP trips off and causes asymmetric flow in the reactor. This will cause diverging cold leg temperatures and mass flow rates due to the differing amounts of time the fluid spends in each leg’s respective steam generator. Consistent with the assumptions for the automatic reactor protective trips, a scram occurs after two seconds.

Scenario D

This transient involves one RCP experiencing a locked motor where the ∆P immediately drops to 0 and the k-losses associated with the pump instantly skyrocket as the pump is unable to turn in any direction. This condition effectively eliminates all flow going through that loop. Consistent with the assumptions for the automatic reactor protective trips, a scram occurs after two seconds.

Scenario E

This transient involves one RCP undergoing a broken shaft. In this condition, while the immediate ∆P drops to 0, the broken shaft allows flow to pass through the pump in either direction. Reverse flow would be expected in this condition, as the other three RCPs would provide a significant diving head into the bottom of the reactor vessel which could cause the flow to enter in the affected loop’s cold leg. Consistent with the assumptions for the automatic reactor protective trips, a scram occurs after two seconds.

Scenario F

This scenario analyzes more prototypic conditions which could cause asymmetric loop behavior – plugged steam generator tubes. This scenario seeks to determine the number of plugged tubes required to reduce the flow rate in the impacted loop by 10%. As the number of steam generator tubes factor into the equivalent steam generator heat transfer area, the Colebrook friction factor, and the minor k-losses, reducing the number of tubes reduces these factors which increase the overall flow resistance in the loop.

# Results and Discussion

Scenario A

A graph of a curve

Description automatically generatedTo meet the required mass flux, the RCP required a ∆P rated of 5560 psig. Clearly, this number is too large to be realistic for a standard RCP used in PWR applications. However, this value was used for the remainder of the transients as the corresponding mass flow rate and pressure correlations were satisfied when providing this rated ∆P. The corresponding flow rate can be seen in the figure:

Scenario B

In this scenario, there are two major concerns. The first involves an immediate temperature rise after the loss of flow event and before the reactor scrams. The reduction in mass flow from the tripped pump causes the coolant to spend more time in the core, greatly increasing the core outlet temperature. Similarly, the hotter coolant increases the PCT calculated in the four core nodes. The goal of this transient was to determine the minimum β value required to ensure that both the core exit and peak cladding temperature do not exceed saturation temperature. The larger the beta, the slower the pump will coast down, the higher the flow rate before the scram, and so the lower the total temperature rise. Below are the graphs of core exit temperature, clad temperature, and mass flow:

A graph of different colored lines

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A graph of a line

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Here, the model behaves as expected. Once the pumps trip off, both the core exit temperatures and the cladding surface temperature begin to rise until the point the reactor is scrammed two seconds following the trip. Of note is the third core node has the highest (most limiting) cladding temperature. This follows from the given power peaking factors and the temperature rise as the fluid rises in the core. The power peaking factor for nodes two and three are greater than nodes one and four. The bulk fluid temperature increases axially through the core due to the addition of heat from the core nodes, and so while the bulk fluid temperature is higher in node 4, the large power peaking factor in node 3 causes this clad temperature to be more limiting.

The minimum required β to ensure that the cladding surface temperature stayed below saturation temperature was 3.8.

Scenario C

A graph of a line graph

Description automatically generatedThis scenario is similar to Scenario B, but only the RCP in loop one trips offline. This is the first scenario which results in asymmetric loop conditions through the core, but the overall concerns remain the same as in Scenario B. As only one RCP trips off, the overall mass flow will not decrease across the core which will prevent as severe a heatup before the scram. Consistent with this discussion, the peak temperature for the core exit and the peak cladding temperature does not approach the same peaks as in Scenario B. And while the flow rate for the loop with the tripped RCP decreases, it is important to note the sudden jump in flow rate after the reactor scram occurs. This could be a result of lower friction factors and k-losses from the drop in heat transfer across the core lowering the ∆P across the reactor core, and the scale of the below mass flow rate graph is adjusted to show this impact.

A graph of a function

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A graph of a line

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A graph of a line

Description automatically generated

Scenario D

In this transient, the RCP for loop one immediately experiences a locked motor event and prevents any flow through the loop, resulting in the mass flow shifting to the other three symmetric loops. The steam generators are unable to keep up with this sudden increase in internal energy, which drives up the cold leg temperature of the three loops. However, as the scram and decay heat cause the reactor to significantly reduce the amount of heat generated in the core, the lower heat generation and continued heat removal from the remaining steam generators results in cold leg temperature decreasing again approximately 2.5 seconds after the scram occurs.

A graph of a pump locked rotor

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Description automatically generatedA graph of a function

Description automatically generatedA graph with a line and numbers

Description automatically generated with medium confidenceA graph of a diagram

Description automatically generated

Scenario E

This transient was unable to be fully modeled with the current state of the Python simulation. Even once the RCP pump shear occurred to allow reverse flow conditions, the calculated pressure drop across the core was still low enough that the flow preferred to travel up the reactor vessel as opposed to around the downcomer region associated with loop one. This clearly illustrates an error in the implementation of the momentum balancing and internal energy equations. This behavior was seen with both the iterative and Newton-Raphson solvers. While the mass flow rate does not appropriately update to reflect the reverse flow through loop one, the rest of the PWR behaves similar to Scenario C, where there is an initial decrease in flow through the impacted loop. However, this transient sees the mass flow rate increase by a greater proportion after the scram, most likely as a result of the exit loss coefficient of the pump providing less friction. Below are the graphs showcasing the plant performance, where again the outlet temperature peaks for the symmetric three loops initially until decaying away consistent with the discussion in Scenario D. In all scenarios, the peak cladding temperature for node four becomes the greatest immediately following the scram. Consistent with the discussion for why the node three cladding temperature was limiting while the reactor was at power, as soon as the heat generation in node three decreases as a result of the scram, the greater fluid temperature at node four causes the overall cladding temperature to overtake that at node three. A graph of a line

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Description automatically generatedThe graphs showcasing plant performance are below:

A graph of a function

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If the model showcased reverse flow, then the expected plant response would be considerably more dynamic than shown in this scenario. A reverse flow condition would result in less flow going through the reactor vessel. As there is less flow through the reactor for the same heat flux, the fluid that is traveling through the core would heat up significantly more than the nominal pump trip conditions previously analyzed.

Scenario F

A graph of a line

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Description automatically generatedPlugging steam generator tubes causes a reduction in heat transfer area, an increase in k-losses and greater Colebrook friction factors. In order to reduce the total mass flow rate to 10% of the nominal loop one flow rate, 1028 steam generator tubes had to be clogged, approximately a 84.5% reduction from the nominal number of steam generator tubes. This caused an associated increase in the cold leg temperatures for all four loops, however the cold leg temperature for the loop with clogged tubes increased more (1°F vice 0.2°F). This follows from the reduction in steam generator heat transfer surface area and the slower mass flow rate causing the fluid to stay in the loop one steam generator for a longer period of time.

This simulation was ran for an additional 10 seconds to observe the final steady state conditions.

# Conclusions

This report successfully modeled a four loop PWR with a single pass core in order to determine appropriate design parameters and analyze the impacts of specific transients impactful to reactor safety analysis. Reactor safety was demonstrated for a loss of all AC casualty where all RCPs trip off, a single RCP trip, and a single RCP locked motor casualty. The model was unable to successfully demonstrate acceptable performance for the RCP shaft shear condition as it could not produce reverse flow in the core. As a result, the analysis performed for that particular scenario is not indicative of real plant performance.

As a whole, the model responded in predictable ways expected from a four loop PWR to the various transients, however there were several issues with the model. As discussed, the implementation of the internal energy and momentum balancing equations did not permit reverse flow to occur. Additionally, the calculated ∆P Rated for the RCP was significantly higher than expected. While this value enabled the PWR to function as expected, this could further point to an error in the implementation of the momentum equations as part of the Python model.

There were additional problems in determining the proper tradeoff between the steam generator fudge factor and the required mass flow rates. A significant amount of effort went into determining the appropriate value. Many runs would fail as a result of fluid internal energies going out of the single phase bounds if the steam generator was either pulling too much heat from the fluid or not removing enough compared to the reactor heat generation term.

This model focused on single phase flow, but future work could expand the internal energy, flow resistance, and heat transfer equations to account for two phase flow.

There were additional analyses significant to reactor safety which were not performed as part of this report. Long term decay heat removal was not discussed or modeled due to the focus on short term plant responses to the transient and automatic protective actions, but this can lead to thermal limit violations as a result of exceeding peak centerline temperatures.

# References

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